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Effect of NUCLEAR RADIATION on Materials at CRYOGENIC TEMPERATURES

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JULY 1964-
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LOCKHEED NUCLEAR PRODUCTS

ER-7604

QUARTERLY PROGRESS REPORT No. 15

July 1964 through September 1964

Effect of Nuclear Radiation on materials at Cryogenic Temperatures

PREPARED UNDER

National Aeronautics/Space Administration
Contract NASw-114

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NASA CRYOGENICS
PROJECT MANAGER

Lockheed-Georgia Company

LOCKHEED
NUCLEAR
PRODUCTS

A Division of Lockheed Aircraft Corporation

Contract: NASw-114

Code: U

FOREWORD

This quarterly report is submitted to the Office of Space Launch Vehicles of the National Aeronautics and Space Administration in accordance with the requirements of NASA Contract NASw-114.

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1 INTRODUCTION

This report describes progress made on Contract NASw-114 during the third quarter, July through September 1964.

Screening tests continued on materials, both in-pile and out-of-pile. Post-irradiation metallographic analyses were conducted on a number of samples that had been tested and the mechanical properties of which have been previously reported.[1] The test results of the alloys completed thus far and the discussion of metallographic examination analyses of materials so far examined are contained in the Test Program Section (Section 4) of this report.

During the period covered by this report, thirty (30) smooth and twenty-two (22) notched tensile specimens were irradiated at 30^{o}R. Four (4) of these specimens were lost due to loss of specimen temperature control, two (2) were lost due to irrecoverable reactor scrams and one (1) was lost due to ice formation inside the test chamber of the loop. During the testing of two samples, both 17-7 PH Stainless Steel, the extensometer failed to record strain properly through the full extent of elastic behaviour.

In general, the project equipment performed satisfactorily during the reporting period with maintenance work being performed during the periods in which the reactor was not operating. Repairs were undertaken on two of the test loops. One loop was damaged during a preceeding reporting period; the other required re-work of instrumentation leads as described in Section 2.1.2 of this report. The refrigeration system expansion engines were disassembled, cleaned and rebuilt following a test aborted due to an expansion engine failure described in Section 2.4 of this report.

A new design was completed for transfer of the cask through the airlock penetration of the containment vessel. Drawings of the new design proposed were completed and are being submitted to NASA for approval.

(1). Quarterly Progress Report Nos. 13 and 14, ER-6929 and ER-7352.

2 EQUIPMENT

2.1 TEST LOOPS

2.1.1 Repair of Test Loop 201-003

NASA approval was obtained for a procedure for the repair of the indentation in Test Loop 201-003 which resulted from the accidental closure of the beam port valve on the loop while still inserted in the beam port.(2)

The procedure provided for the transfer of Test Loop 201-003 from the containment vessel to the Hot Materials Handling Area in a manner similar to that used earlier in the transfer of Loop 201-002.(3) Following this transfer of the test loop, the repair of the indentation was instituted following the approved procedure; however, in the process of removing the indentation, the out-of-round and axial misalignment were not remedied. The procedure and required tooling are being developed for the correction of these conditions. It is expected that further specific remedial action will be taken during the next quarter.

2.1.2 Test Loop 201-004 - Extensometer Leads

During an in-pile exposure in Reactor Cycle 21P, a quantity of water entrapped in the test chamber was converted by the helium stream to low temperature, high strength ice. This caused the loss of a test and, during head removal following this incident, the extensometer leads in test loop 201004 were damaged at the front bulkhead. Approval was received from NASA for a procedure to replace these leads with the loop in the Hot Cave and the quadrant drained. Following removal of the head assembly from the front end of the test loop and the instrument tower from the aft end of the test loop, the instrument lines were disconnected. The forward bulkhead feed-throughs were removed by melting the silver solder holding them in place. After they were removed, the bore of the instrument tubes was cleaned thoroughly. New lines were made up and installed by reversing the above procedure and secured to new feed-through connectors with silver solder. The instrument tubes were evacuated from the aft end of the test loop and checked for leakage with a mass spectrometer helium leak detector located beside the test loop in Quadrant "D".

While the loop was open at the aft end for check-out of the repair work, Quadrant "D" was inadvertently flooded causing damage to the helium leak

(2). Quarterly Progress Report No. 13, ER-6929, Page 4.
(3). Quarterly Progress Report No. 14, ER-7352, Pages 3 and 4.

detector and requiring some rework of the test loop. The water entered the tubes carrying the instrument lines, necessitating removal of the water and subsequent drying of the lines and tubes. This incident occurred when water was being transferred from Quadrant "A" to Canal "E", by the NASA Reactor Operations Shift Team. Authority was immediately given to repair the leak detector and the test loop. This was accomplished in time to avoid delay in scheduling the next reactor cycle. The test loop has performed satisfactorily in subsequent cycles.

2.1.3 Extensometer Operation

Of the thirty (30) unnotched tensile specimens tested in-pile during this reporting period, the extensometer failed to record strain in the upper limit of elastic behaviour in two instances. Both samples involved were 17-7 PH Stainless Steel. The tests provide accurate ultimate tensile strength and ductibility data, but the yield strength was not recorded. If further testing shows that the shape of the stress-strain curve for this alloy tested at 30^{o}R after irradiation at 30^{o}R does not have an effective 0.2% offset yield point, due to failure prior to 0.002 in/in plastic strain, the results of these two tests will be included in the screening program test data. Otherwise, the tests will be re-run to provide accurate tensile yield strength data for this material.

2.1.4 Test Loop Head Assemblies

A small but significant reduction in the externally applied heat load required to balance the refrigerator system for specimen temperature control has been observed during the past several months. This indicates that, after one year of intermittent in-pile exposure but approximately two and one-half years after initial evacuation, the annular space in the test loop heads no longer has a sufficiently high vacuum to maintain the initial insulating capacity. Since further deterioration of the heads could possibly result in inability to maintain proper specimen temperature control, an investigation of the possibility of re-evacuating and sealing this area on irradiated, and therefore radioactive, heads is being undertaken.

2.2 REMOTE HANDLING EQUIPMENT AND SAMPLE CHANGE SYSTEM

2.2.1 Carriages

One bearing failure occurred during this period in the carriage test loop drive mechanism which had been modified as previously described.[4]

(4). Quarterly Progress Report No. 14, ER-7352, Page 5.

The initial indication that a failure had occurred was an increase in the time required for a test loop insertion into HB-2 against primary coolant pressure. The irradiation and testing of the specimen in the test loop at the time was completed without incident. Upon completion of the test, the test loop was transferred from the affected carriage (No. 4) and the carriage removed from Quadrant "D" for disassembly and inspection.

Upon disassembly, inspection of the bearings showed no evident damage to the 18 Ni (300) Maraging Steel outer races. The substitution of this material for AISI 440C Stainless Steel had been the principal modification [4] to reduce the incidence of bearing failure. However, two of the AISI 440C inner races were found to be cracked and the remaining bearings were partially frozen by accumulated sedimentation deposit from the quadrant water. There was evidence of slight wear on the needles causing an out-of-round needle configuration. All eight bearing assemblies were cleaned and reassembled using new needles and replacements for the broken inner races. No design modification is contemplated as a result of this incident other than the increase in time required to insert the test loop into HB-2. No operational delay was incurred and the condition is considered to be a matter of routine maintenance rather than one requiring remedial action.

2.2.1.1 Clevite 10 HP Pump

The 10 HP Clevite pump, modified during the proceeding quarter [5] to eliminate admixture of the deionized water hydraulic fluid with the lubricating oil, has been operated for an additional five (5) reactor power cycles without difficulty.

2.2.2 Air Lock Penetration

The activity on the modification of the air lock, bridge and cart to facilitate transfer of the test loop cask through the air lock penetration of the containment vessel [6] was resumed during this reporting period. The present design concept increases the adaptability of the air lock facility for experiments other than the Lockheed test loop transfer cask. The new design is complete and will be submitted to NASA for approval during the next reporting period.

2.2.3 Hot Cave Valve

Excessive leakage of quadrant water into the Hot Cave was observed during this period. During 24S down cycle, when Quadrant "D" was drained, the

(4). Quarterly Progress Report No. 14, ER-7352, Page 5.
(5). Quarterly Progress Report No. 14, ER-7352, Page 6.
(6). Quarterly Progress Report No. 13, ER-6929, Page 14.

three teflon chevron seals in the Hot Cave Port Valve were removed for inspection. All three seals were cracked. These seals were replaced with six (6) polyurethane seals making the installation similar to the reactor beam port seal configuration. The greater flexibility of the polyurethane materials is expected to provide increased seal life as well as improved sealing capability. Approximately fifteen (15) test loop insertions into the Hot Cave were made during Cycle 24P without incident or evidence of excessive leakage of quadrant water into the Hot Cave drain.

2.2.4 Beam Port Valve

A slight deterioration of the sealing action of the polyurethane chevron seals in the reactor beam port coupling assemblv was indicated late in Reactor Cycle 23P by a rise in the auxiliary seal water flow. The seals were removed during Cycle 24S and damage to the lip section was observed. These seals were installed during Reactor Cycle 11S in January 1964 and have withstood more than one hundred and fifty (150) test loop insertions since that time. Therefore, the wear pattern is considered to be normal and the seals have provided adequate service life. The damaged seals were replaced and the seal assembly functioned without incident for approximately ten (10) test loop insertions in Cycle 24P.

2.3 BEAM PORT SHIELD

The build-up of activity in the primary coolant water from W^{187} continues to constitute an operational problem[7] for the NASA Reactor Operations personnel. Planning of methods to rectify this condition continued during this reporting period, but no active work was undertaken.

2.4 REFRIGERATION SYSTEM

During the period covered by this report, fifty-two (52) specimens were irradiated at $30^{\circ}R$. In four (4) instances tests were aborted before the specimen had received the required 1×10^{17} nvt exposure due to loss of specimen temperature control.

The first aborted test was caused by a refrigeration system shut-down due to the bursting of a 1" aluminum 350 psi rupture disc (V67) located on the high pressure side of the refrigeration system in the containment vessel. Examination of the failed rupture disc disclosed extensive plastic flow

(7). Quarterly Progress Report No. 14, ER-7352, Pages 6-9.

prior to failure. Similar plastic flow has been observed on used but un-failed rupture discs and is probably caused by the cyclic pressure variation in the system due to the action of the compressor. This condition would reduce the effective thickness of the rupture disc and reduce its bursting pressure to below rated capacity. To prevent re-occurrence of this type of failure, a design change to include a 1" relief valve set at 345 psi in parallel with a 1/2" aluminum 365 psi rupture disc at this point was submitted and approved by NASA. This will utilize the re-seating relief valve as a primary safety device with the rupture disc serving as a positive back-up precaution in the event that the relief valve fails to operate. The rupture disc will be inspected at frequent intervals and replaced if plastic deformation is observable. The installation of this change is anticipated in the ensuing quarter.

The second test on which temperature control could not be maintained occurred through the loss of insulating vacuum in the annular space of the transfer lines. The lines were subsequently re-evaluated and have been used successfully since the incident.

The third aborted test occurred because of an unscheduled compressor stoppage caused by a blown fuse in the 1 KV transformer circuit of the compressor. After fuse replacement, the compressor operated satisfactorily. Measurement of the circuit current during normal operating conditions showed that the proper power was being used and the fuse failure is attributed to a transient condition in the circuit.

The final aborted test in this reporting period also was caused by refrigeration system shut-down. This incident was caused by the failure of a piston rod in the expansion engine in engine location number 1. This failure was of a type similar to those encountered in previous periods. [8] The cause of failure has yet to be determined. The piston rods of all four operating expansion engines were replaced with new rods. The spare engines, as well as the operating system, was completely overhauled, cleaned and rebuilt. Since the failure occurred during Cycle 24P, the final power cycle of this reporting period, no operating experience with the rebuilt expansion engine was obtained. Since the most recent piston rod failure prior to that of Reactor Cycle 24P occurred during Reactor Cycle 15S in March 1964, this type of failure appears to be an isolated, infrequent occurrence.

It was observed that the paint was discolored and peeling from the first stage of the Ingersoll-Rand Compressor due to the heat build-up in this area during continuous operation. All of the remaining paint was removed

(8). Quarterly Progress Report No. 13, ER-6929, Page 16.

from the Compressor and it was repainted with a heat resisting paint warranted to withstand 400 ^{o}F without discoloration and 700 ^{o}F without blistering.

Except as noted above, the refrigeration system performed satisfactorily during the reporting period with no servicing other than routine maintenance.

3 FLUX MAPPING

During this reporting period, which included Reactor Power Cycles 20P, 21P, 22P, 23P and 24P, the reactor loading was changed from 168 gram fuel elements to 200 gram elements. The loading change was made over several cycles. In order to determine any change in fast neutron flux caused by the variations in lattice loading, four (4) sets of foils were run during this quarter. The method used for flux measurements has been previously described.[9] Since both flux level and spectral shape were of interest, complete sets of foils were run in each case.

In no case was there a significant variation in fast flux level or spectral shape from prior data obtained with similar reactor parameters. Data obtained was in agreement, within normal experimental variation, with the published flux curves.[10] No changes in exposure duration or method of its calculation were indicated by the flux measurements made during this reporting period.

For the final three foil exposures of this quarter, parallel sets of foils were irradiated for evaluation by NASA personnel at Plum Brook and by Lockheed personnel at Dawsonville, Georgia. The NASA supplied foils are smaller than the Lockheed foils, and NASA substituted a U^{238} foil with an activation energy of 1.45 Mev. for the Lockheed Th^{232} foil with an activation energy of 1.75 Mev. Agreement between the spectral curves plotted from the similar sets of foils is well within the established limits of experimental uncertainty and it is anticipated that future flux mapping requirements will be met using foil evaluation at Plum Brook.

The NASA Nuclear Experiments Section supplied pure Cobalt foils for irradiation to monitor the thermal flux in the test location in HB-2. These foils were irradiated together with one set of the fast neutron monitoring foils described above. The foil exposure was performed in Reactor Power Cycle 21P at a power level of 60 MW and the bank of fueled control rods advanced to 25 inches. Under these reactor operational conditions the thermal flux was found to be in the order of 1×10^4 n/cm^2/watt/sec, or approximately 28% of the fast flux.

(9). Quarterly Progress Report No. 12, ER-6793, Pages 15-22.
Quarterly Progress Report No. 11, ER-6590, Pages 21-25.

(10). Quarterly Progress Report No. 13, ER-6929, Figures 5 and 6, Pages 18 and 19.

4 TESTING PROGRAM

4.1 SCREENING TESTS

The in-pile portion of the screening program was conducted in the five reactor power cycles, 20P through 24P, which occurred during this reporting period. Out-of-pile testing was also performed during reactor down cycles when equipment maintenance schedules permitted operation of the refrigeration system.

During this period, at the conclusion of reactor cycle 23P on 24 August 1964, the testing of tensile notch specimens was temporarily suspended to permit more rapid generation of tensile test data. This was done at the request of Lewis Research Center personnel.

The in-pile testing of three specimen lots of all the Nickel alloys included in the screening program was completed during this period, and the test results are given in Tables 1 through 4. Out-of-pile test results for the Nickel alloys with the exception of Inconel X were previously reported.(11) Out-of-pile room temperature test data on Inconel X were previously reported(12) for a three specimen sample lot. Additional tests to increase the sample size to five specimens did not affect the reported test results. The test results of Inconel X specimens tested at 30°R unirradiated are given in Table 5.

Testing tensile notch specimens under these conditions had not been completed at the end of the reporting period. However, since the notched tensile/unnotched tensile ratio at 30°R after 1×10^{17} nvt fast neutron exposure is essentially unity, excessive notch sensitivity would not be likely to appear as a result of cryogenic environment alone.

A summary of all test results for each of the Nickel alloys is given in Tables 6 through 9.

Screening testing of Steel alloys was also conducted during this reporting period. In-pile testing was completed on Type A-286 Stainless (AMS5737), Type AM350 Stainless (SCT) and Type 440C Stainless (Rc 60). The test data are given in Tables 10 through 12. Previously unreported out-of-pile test data is given in Tables 13 through 18. Tensile notch testing of 440C out-of-pile had not been completed at the end of the reporting period; so only partial results are reported.

(11). Quarterly Progress Report No. 9, ER-6219, Tables 23 through 27.
(12). Quarterly Progress Report No. 13, ER-6929, Table 3.

A summary of all test results for each of the three alloys is given in Tables 20 and 22. To facilitate direct comparison between the test results obtained from A-286 in different heat treating conditions, Table 19 is reproduced from previously reported material.[13]

A significant irradiation induced embrittlement is observable in AM 350 Steel, evidenced particularly by the deterioration of the notched-unnotched ratio after irradiation and verified by the reduction in magnitude of the ductility parameters.

A measurable reduction in tensile strength was observed in 440C Steel as a result of irradiation. The reported increase in notched-unnotched ratio is a result of impaired tensile strength rather than a significant change in notch sensitivity. The almost complete absence of ductile behaviour of this material at cryogenic temperatures make interpretation of the tri-axial loading pattern at the root of the notch of questionable validity. Conclusions based on quantative variation of notched-unnotched ratios below 0.50 are of dubious value for brittle materials.

4.2 PRELIMINARY METALLOGRAPHIC STUDIES

Some preliminary metallographic studies have been undertaken. Metallographic studies of failed tensile specimens from which tensile results were reported earlier[14] were undertaken in an attempt to confirm the unusual tensile test results and to determine the general usefulness of this technique in conjunction with tensile testing under nuclear cryogenic conditions.

Although limited in their usefulness, metallographic studies of failed unnotched tensile specimens might help to understand effects due to irradiation at low temperatures, which occur in tensile strength, elongation, notch strength of cold worked material, reduction in area and fracture stress.

When such effects due to radiation occur during tensile testing, metallographic examination might be expected to show that the results are confirmed by differences in degree of grain deformation (particularly in necked down regions where there is a large difference in reduction in area) and by the type of fracture (whether intergranular or trans-granular of either cleavage or non-cleavage type). Also, it may be

(13). Quarterly Progress Report No. 14, ER-7352, Page 37.

(14). Quarterly Progress Report No. 13, Page 21.
Quarterly Progress Report No. 14, Page 15.

TABLE 1 TEST RESULTS, RENE 41, TEST TEMPERATURE: 30° R

TENSILE TEST						
Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation In 1/2" (4 Diameters), %	Reduction Of Area, %	Fracture Stress psi
7 Ca 8	1×10^{17}	196,200	113,600	54.2	48.7	381,600
7 Ca 36	1×10^{17}	192,300	109,900	Not Recorded	Not Recorded	Not Recorded
7 Ca 40	1×10^{17}	188,800	111,000	50.8	36	313,000
Average	-	192,400	111,500	52.5	42.4	347,300
Scatter	-	$\pm 2\%$	+ 1.9%, -1.4%	$\pm$ 3%	$\pm$ 15%	$\pm$ 9.9%

TENSILE NOTCH TEST			
Specimen Number	Total Accumulated Fast Neutron Dose, (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Notched-Unnotched Ratio
7 Ca 29	1×10^{17}	193,600	Avg ÷ Avg 0.997
7 Ca 31	1×10^{17}	199,100	Low ÷ High 0.959
7 Ca 37	1×10^{17}	187,800	High ÷ Low 1.026
Average		193,500	
Scatter		$\pm 3\%$	

TABLE 2 TEST RESULTS, K MONEL, TEST TEMPERATURE: 30° R

TENSILE TEST

Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation In 1/2" (4 Diameters), %	Reduction Of Area %	Fracture Stress psi
1 Fb 12	1×10^{17}	188,000	141,600	32.8	47.8	317,800
1 Fb 30	1×10^{17}	190,000	138,600	32.2	48.1	345,000
1 Fb 38	1×10^{17}	187,200	136,600	32.0	52.2	295,100
Average		188,400	138,900	32.3	49.4	319,300
Scatter		+ .8%, -.6%	+ 1.9%, -1.6%	+ 1.5%, -.9%	+ 6%, -3%	$\pm$ 8%

TENSILE NOTCH TEST

Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Notched-Unnotched Ratio	
1 Fb 27	1×10^{17}	207,300	Avg ÷ Avg	1.12
1 Fb 40	1×10^{17}	207,900	Low ÷ High	1.09
1 Fb 45	1×10^{17}	216,000	High ÷ Low	1.15
Average		210,400		
Scatter		+ 2.6%, -1.5%		

TABLE 3 TEST RESULTS, INCONEL, TEST TEMPERATURE: 30° R.

TENSILE TEST

Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation In 1/2" (4 Diameters), %	Reduction Of Area %	Fracture Stress psi
2 Fb 68	1×10^{17}	188,000	177,000	26.5	54.3	387,700
2 Fb 69	1×10^{17}	194,000	180,500	26.2	52.9	350,400
2 Fb 84	1×10^{17}	182,000	175,300	23.5	54.9	349,500
Average		188,000	177,600	25.4	54	362,500
Scatter		$\pm$ 3.2%	+ 1.6%, -1.3%	+ 4.3%, -7.5%	+1.7%, -2.0%	+ 7%, -3.6%

TENSILE NOTCH TEST

Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Notched-Unnotched Ratio	
2 Fb 11	1×10^{17}	235,000	Avg ÷ Avg	1.25
2 Fb 14	1×10^{17}	240,000	Low ÷ High	1.18
2 Fb 37	1×10^{17}	229,650	High ÷ Low	1.32
Average		234,900		
Scatter		$\pm$ 2%		

TABLE 4 TEST RESULTS, INCONEL X, PRECIPITATION HARDENED, TEST TEMPERATURE: 30° R

TENSILE TEST						
Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation In 1/2" (4 Diameters), %	Reduction Of Area, %	Fracture Stress psi
3 Fa 22	1×10^{17}	236,700	159,300	29.4	46.0	426,200
3 Fa 38	1×10^{17}	235,000	152,000	28.8	42.0	390,000
3 Fa 42	1×10^{17}	234,000	161,000	28.0	32.0	344,200
Average		235,200	157,400	28.7	40.0	386,800
Scatter		+ .6%, - .5%	+ 2.3%, -3.4%	$\pm$ 2.4%	+ 15%, -20%	$\pm$ 11%

TENSILE NOTCH TEST			
Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Notched-Unnotched Ratio
3 Fa 45	1×10^{17}	230,000	Avg ÷ Avg 0.98
3 Fa 52	1×10^{17}	229,700	Low ÷ High 0.96
3 Fa 55	1×10^{17}	229,650	High ÷ Low 0.98
Average		229,800	
Scatter		$\pm$ 0.1%	

TABLE 5 OUT-OF-PILE TEST RESULTS, NICKEL ALLOYS, TEST TEMPERATURE: $30^{\circ}R$
ALLOY: INCONEL X - PRECIPITATION HARDENED

TENSILE TEST					
Specimen Number	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation In 1/2" (4 Diameters), %	Reduction Of Area %	Fracture Stress psi
3 Fa 5	252,600	158,000	32	44	430,000
3 Fa 10	245,000	154,000	31	45	406,000
3 Fa 14	248,500	157,000	31	46	450,000
3 Fa 17	253,000	160,700	31	49	450,000
3 Fa 19	242,000	146,500	31	46	433,000
Average	248,200	155,200	31	46	433,800
Scatter	+ 1.7%, -2.5%	+ 3.5%, -5.6%	NIL	+ 4.3%, -6.5%	+ 3.7%, -6.4%

TABLE 6 SUMMARY OF TEST RESULTS, RENE 41

PROPERTY	TEST CONDITIONS				
	ROOM TEMP, * UNIRRADIATED	30° R, UNIRRADIATED *		30° R, IRRADIATED TO 1×10^{17} nvt FAST NEUTRONS	
	Test Data, Average of 5 Tests	Test Data, Average of 5 Tests	Net Change Due to Temperature	Test Data, Average of 3 Tests	Net Change Due to Irradiation
Ultimate Tensile Strength, F_{tu}, psi	129,300	194,100	+50%	192,400	- 0.9% NIL
Tensile Yield Strength, F_{ty}, psi	63,200	107,600	+70%	111,500	+ 4% NIL
F_{ty}/F_{tu} Ratio	0.49	0.55	+12%	0.58	+5.5% NIL
Tensile Notch Strength (K_t = 6) psi	142,900	204,600	+43%	193,500	- .5% NIL
Notched-Unotched Ratio	1.11	1.05	- 5.4%	1.00	- 5% NIL
Fracture Stress	Not Recorded	Not Recorded	-	347,300	-
Elongation in 1/2" (4 Diameters), %	56	62	+10.7%	52.5	-15.3%
Reduction of Area, %	59	63	+6.8%	42.4	-32.7%

* Test Results Reported in Quarterly Progress Report No. 9, ER 6219.

TABLE 7 SUMMARY OF TEST RESULTS, K MONEL

PROPERTY	TEST CONDITIONS				
	ROOM TEMP, * UNIRRADIATED	30° R, UNIRRADIATED *		30° R, IRRADIATED TO 1×10^{17} nvt FAST NEUTRONS	
	Test Data, Average of 5 Tests	Test Data, Average of 5 Tests	Net Change Due to Temperature	Test Data, Average of 3 Tests	Net Change Due to Irradiation
Ultimate Tensile Strength, F_{tu}, psi	154, 200	189, 500	+22.9%	188, 400	-0.6% NIL
Tensile Yield Strength, F_{ty}, psi	97, 400	123, 100	+ 26.4%	138, 900	+ 12.8%
F_{ty} / F_{tu} Ratio	0.63	0.65	+ 3.2% NIL	0.74	+ 13.8%
Tensile Notch Strength (K_t =6 psi)	180, 300	209, 800	+ 16.4%	210, 400	+ 2.9% NIL
Notched-Unnotched Ratio	1.17	1.12	-4% NIL	1.12	NIL
Fracture Stress	Not Recorded	Not Recorded	---	319, 300	---
Elongation in 1/2" (4 Diameters), %	26	34	+ 31%	32.3	-5% NIL
Reduction of Area, %	47	52	+ 10.6% NIL	49.4	-5% NIL

*Test Results Reported in Quarterly Progress Report No. 9, ER 6219.

TABLE 8 SUMMARY OF TEST RESULTS, INCONEL

PROPERTY	TEST CONDITIONS				
	ROOM TEMP, * UNIRRADIATED	30° R, UNIRRADIATED *		30° R, IRRADIATED TO 1 x 10^{17} nvt FAST NEUTRONS	
	Test Data, Average of 5 Tests	Test Data, Average of 5 Tests	Net Change Due to Temperature	Test Data, Average of 3 Tests	Net Change Due to Irradiation
Ultimate Tensile Strength, F_{tu}, psi	138, 100	186, 300	+ 35%	188, 000	+ 0.9% NIL
Tensile Yield Strength, F_{ty}, psi	129, 600	175, 000	+ 35%	177, 600	+ 1.5% NIL
F_{ty} / F_{tu} Ratio	0.94	0.94	NIL	0.94	NIL
Tensile Notch Strength (K_t = 6) psi	178, 900	222, 700	+ 24%	234, 900	+ 5.5%
Notched-Unnotched Ratio	1.30	1.20	-7.7%	1.25	+ 4.2% NIL
Fracture Stress	Not Recorded	Not Recorded	--	362, 500	
Elongation in 1/2" (4 Diameters), %	27	26	-3.7% NIL	25.4	-2.3% NIL
Reduction of Area, %	55.0	54.0	-1.8% NIL	54.0	NIL

* Test Results Reported in Quarterly Progress Report No. 9, ER 6219.

TABLE 9 SUMMARY OF TEST RESULTS, INCONEL X

PROPERTY	TEST CONDITIONS				
	ROOM TEMP,* UNIRRADIATED	30° R, UNIRRADIATED		30° R, IRRADIATED TO 1×10^{17} nvt FAST NEUTRONS	
	Test Data, Average of 5 Tests	Test Data, Average of 5 Tests	Net Change Due to Temperature	Test Data, Average of 3 Tests	Net Change Due to Irradiation
Ultimate Tensile Strength, F_{tu}, psi	201,000	248,200	+ 23.5%	235,200	-5% NIL
Tensile Yield Strength, F_{ty}, psi	141,300	155,200	+ 9.8%	157,400	+ 1.4% NIL
F_{ty}/F_{tu} Ratio	0.70	0.63	- 10%	0.67	+ 6.3%
Tensile Notch Strength ($K_t = 6$) psi	245,700	Not Recorded	-	229,800	-
Notched-Unnotched Ratio	1.22	Not Recorded	-	.98	-
Fracture Stress	Not Recorded	433,800	-	386,800	-11%
Elongation in 1/2" (4 Diameters), %	24	31	+ 29%	28.7	-7.4%
Reduction of Area, %	40	46	+ 15%	40	-13%

*Test Results Reported in Quarterly Progress Report #13, ER 6929.

TABLE 10 TEST RESULTS, A-286 STEEL, AMS 5737, TEST TEMPERATURE: 30° R

TENSILE TEST						
Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation In 1/2" (4 Diameters), %	Reduction Of Area %	Fracture Stress psi
8 Ca 5	1×10^{17}	230,000	160,700	32	46	310,200
8 Ca 13	1×10^{17}	231,000	163,700	30	42	398,000
8 Ca 19	1×10^{17}	235,000	164,400	24	39	355,000
Average		232,000	162,900	29	42	354,400
Scatter		+ 1.2%, -.9%	+ .9%, -1.4%	+ 10%, -17%	+ 10%, -7%	± 12%

TENSILE NOTCH TEST			
Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Notched-Unnotched Ratio
8 Ca 25	1×10^{17}	233,700	Avg ÷ Avg 1.00
8 Ca 39	1×10^{17}	238,400	Low ÷ High 0.96
8 Ca 40	1×10^{17}	226,200	High ÷ Low 1.04
Average		232,800	
Scatter		+ 2.4%, -2.8%	

TABLE 11 TEST RESULTS, AM 350 STEEL, SCT, TEST TEMPERATURE: 30° R

TENSILE TEST

Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation In 1/2" (4 Diam-meters), %	Reduction Of Area %	Fracture Stress psi
10 Ca 17	1×10^{17}	305,000	303,650	6.2	18.4	361,000
10 Ca 22	1×10^{17}	330,000	327,000	10.9	29.8	428,200
10 Ca 39	1×10^{17}	305,100	298,700	3.6	13.6	343,800
Average		313,400	309,800	6.9	20.6	377,700
Scatter		+ 5.3%, -2.7%	+5.6%, -3.6%	+ 58%, -48%	+ 45%, -34%	+13%, -9%

TENSILE NOTCH TEST

Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Notched-Unnotched Ratio	
10 Ca 26	1×10^{17}	128,800	Ave ÷ Avg	0.47
10 Ca 28	1×10^{17}	163,600	Low ÷ High	0.39
10 Ca 33	1×10^{17}	153,100	High ÷ Low	0.54
Average		148,500		
Scatter		+10%, - 13%		

TABLE 12 TEST RESULTS, 440C STAINLESS STEEL, TEST TEMPERATURE: 30° R

TENSILE TEST

Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation In 1/2" (4 Diameters), %	Reduction Of Area, %	Fracture Stress psi
9 Ca 18	1×10^{17}	236,100	*	NIL	NIL	**
9 Ca 19	1×10^{17}	222,300	*	NIL	NIL	**
9 Ca 23	1×10^{17}	198,100	*	NIL	NIL	**
Average		218,800	*	NIL	NIL	**
Scatter		+8%, -9.5%				

TENSILE NOTCH TEST

Specimen Number	Total Accumulated Fast Neutron Dose (nvt)	Ultimate Tensile Strength (F_{tu} in psi)	Notched-Unnotched Ratio	
9 Ca 27	1×10^{17}	99,100	Avg ÷ Avg	0.49
9 Ca 31	1×10^{17}	114,200	Low ÷ High	0.42
9 Ca 35	1×10^{17}	107,400	High ÷ Low	0.58
Average		106,900		
Scatter		$\pm$ 7%		

* Specimen failed with less than 0.2% permanent set. Tensile yield equal to ultimate tensile strength.
** Fracture Stress equal to ultimate tensile strength.

TABLE 13 OUT-OF-PILE TEST RESULTS, STEEL ALLOYS, TEST TEMPERATURE: ROOM TEMPERATURE
ALLOY: A-286, AMS 5737 (1650° F SOLUTION TREATED AND AGED)

TENSILE TEST					
Specimen Number	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation in 1/2" (4 Diameters), %	Reduction of Area %	Fracture Stress psi
8 Ca 4*	179,000	141,200	23	50	Not Recorded
8 Ca 7*	174,000	131,400	22	49	Not Recorded
8 Ca 37*	170,000	136,000	24	52	Not Recorded
8 Ca 28	165,800	129,400	24	55	271,000
8 Ca 10	167,900	130,200	23	53	269,200
Average	171,300	133,600	23	52	270,100
Scatter	+4.5%, -3.2%	+5.7%, -3.1%	± 4.3%	± 5.8%	± 0.3%
		TENSILE NOTCH TEST			
8 Ca 9*	182,000				
8 Ca 11*	183,000				
8 Ca 17*	184,500	NOTCHED TO UNNOTCHED RATIO			
8 Ca 27	193,900				
8 Ca 34	192,000	Avg ÷ Avg	1.09		
		Low ÷ High	1.02		
Average	187,100	High ÷ Low	1.17		
Scatter	+3.6%, -2.7%				

*Previously Reported in Quarterly Progress Report No. 13, ER-6929

TABLE 14 OUT-OF-PILE TEST RESULTS, STEEL ALLOYS, TEST TEMPERATURE: 30° R

ALLOY: A-286, AMS 5737 (1650° F SOLUTION HEAT TREATED AND AGED)

TENSILE TEST					
Specimen Number	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation in 1/2" (4 Diameters), %	Reduction of Area %	Fracture Stress psi
8 Ca 8	249,800	164,700	32	44	434,000
8 Ca 12	249,500	163,300	32	45	455,000
8 Ca 18	240,400	158,300	32	46	444,500
8 Ca 20	246,900	167,250	32	45	434,000
8 Ca 21	246,900	154,800	33	43	419,800
Average	246,700	161,700	32	45	437,500
Scatter	+1.3%, -2.6%	+3.4%, -4.3%	NIL	+2.2%, -4.4%	± 4%

		TENSILE NOTCH TEST	
		NOTCHED TO UNNOTCHED RATIO	
8 Ca 29	256,000		
8 Ca 32	250,600		
8 Ca 35	257,200	Avg ÷ Avg	1.03
8 Ca 40	254,500		
8 Ca 41	253,800	Low ÷ High	1.00
Average	254,400	High ÷ Low	1.07
Scatter	+1.1%, -1.5%		

TABLE 15 OUT-OF-PILE TEST RESULTS, STEEL ALLOYS, TEST TEMPERATURE: ROOM TEMPERATURE

ALLOY: AM-350 STEEL, SCT

TENSILE TEST					
Specimen Number	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation in 1/2" (4 Diameters), %	Reduction of Area %	Fracture Stress psi
10 Ca 23*	195, 500	180, 500	14	50	Not Recorded
10 Ca 40*	193, 000	183, 000	17	60	Not Recorded
10 Ca 42*	193, 000	182, 000	12	43	Not Recorded
10 Ca 18	206, 100	192, 000	19	53	310, 200
10 Ca 21	208, 800	192, 900	20	53	318, 500
Average	199, 300	186, 100	16	52	314, 400
Scatter	+4.8%, -3.2%	+3.7%, -3%	$\pm$25%	+ 15%, -17%	$\pm$ 1.3%
		TENSILE NOTCH TEST			
		NOTCHED TO UNNOTCHED RATIO			
10 Ca 1*	250, 000	Avg ÷ Avg	1.29		
10 Ca 12*	278, 000	Low ÷ High	1.20		
10 Ca 14*	253, 000	High ÷ Low	1.44		
10 Ca 20	250, 300				
10 Ca 35	252, 000				
Average	256, 700				
Scatter	+8.3%, -2.6%				

TABLE 16 OUT-OF-PILE TEST RESULTS, STEEL ALLOYS, TEST TEMPERATURE: 30° R

ALLOY: AM-350 STEEL, SCT

TENSILE TEST					
Specimen Number	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation in 1/2" (4 Diameters) %	Reduction of Area %	Fracture Stress psi
10 Ca 9	314,400	311,700	9.6	36	483,300
10 Ca 10	313,900	313,900*	9.6	36	493,000
10 Ca 11	317,000	317,000*	10.8	36	497,900
10 Ca 23	318,400	316,600	10.0	35	502,600
10 Ca 34	311,800	311,800*	9.8	37	475,000
Average	315,100	314,200	9.96	36	490,400
Scatter	± 1%	± 0.8%	+8.4%, -3.69%	± 2.8%	+2.5%, -3%

		TENSILE NOTCH TEST	
		NOTCHED TO UNNOTCHED RATIO	
10 Ca 3	241,300	Avg ÷ Avg	0.85
10 Ca 6	280,600	Low ÷ High	0.76
10 Ca 13	272,600	High ÷ Low	0.90
10 Ca 31	264,100		
10 Ca 36	280,600		
Average	267,800		
Scatter	+ 4.8%, -10%		

* Maximum Stress Level on Stress Strain Diagram Occurred at 0.2% Offset.

TABLE 17 OUT-OF-PILE TEST RESULTS, STEEL ALLOYS, TEST TEMPERATURE: ROOM TEMPERATURE

ALLOY: 440C STAINLESS STEEL

TENSILE TEST					
Specimen Number	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elongation in 1/2" (4 Diameters), %	Reduction of Area %	Fracture Stress psi
9 Ca 1*	322,000	279,000	5	8	350,000
9 Ca 2*	324,000	281,000	5	7	346,000
9 Ca 7*	337,000	303,000	3	6	356,000
9 Ca 8	303,500	255,500	6	10	340,000
9 Ca 13	303,100	245,600	7	12	324,000
Average	317,900	272,800	5.2	8.6	343,200
Scatter	+6%, -5%	+11%, -10%	+35%, -42% NIL	+ 40%, -30% NIL	+3.7%, -5.6%

		TENSILE NOTCH TEST	
		NOTCHED TO UNNOTCHED RATIO	
9 Ca 33*	186,000	Avg ÷ Avg	0.59
9 Ca 34*	189,400	Low ÷ High	0.55
9 Ca 36*	189,000	High ÷ Low	0.62
Additional testing not completed at time of report.			
Average	188,100		
Scatter	+.7%, -1.1%		

* Previously Reported in Quarterly Progress Report No. 13, ER-6929

TABLE 18 OUT-OF-PILE TEST RESULTS, [EEL ALLOYS, TEST TEMPERATURE: 30° R

ALLOY 440C STAINLESS STEEL

TENSIL TEST					
Specimen Number	Ultimate Tensile Strength (F_{tu} in psi)	Tensile Yield Strength (F_{ty} in psi)	Elc gation in 1/2" (4 l ameters), %	Reduction of Area %	Fracture Stress psi
9 Ca 11	286,600	*	NIL	NIL	**
9 Ca 12	246,900	*	NIL	NIL	**
9 Ca 16	276,600	*	NIL	NIL	**
9 Ca 20	240,000	*	NIL	NIL	**
9 Ca 21	248,600	*	NIL	NIL	**
Average	259,700	-	—	—	—
Scatter	+ 10%, -8%	-	—	—	—
		TENSILE NOTCH TEST			
9 Ca 25	103,000	NOTCHED TO UNNOTCHED RATIO		0.40	
Test Data for Tensile Notch Samples Incomplete.					

* Specimen failed prior to 0.2% permanent set - tensile yield equal to ultimate tensile strength.
** Fracture stress equal to ultimate tensile strength.

TABLE 19 SUMMARY OF TEST RESULTS, A-286 STEEL (AMS 5735)

PROPERTY	TEST CONDITIONS				
	ROOM TEMP, UNIRRADIATED	30° R, UNIRRADIATED		30° R, IRRADIATED TO 1×10^{17} nvt FAST NEUTRONS	
	Test Data, Average of 5 Tests	Test Data, Average of 5 Tests	Net Change Due to Temperature	Test Data, Average of 3 Tests	Net Change Due to Irradiation
Ultimate Tensile Strength, F_{tu}, psi	155,900	235,000	+ 51%	229,400	-2.4% NIL
Tensile Yield Strength, F_{ty}, psi	112,000	147,000	+ 31%	155,000	+ 5.4% NIL
F_{ty}/F_{tu} Ratio	0.72	0.63	-12.5%	0.68	+ 8% NIL
Tensile Notch Strength ($K_t = 6$) psi	181,000	216,000	+ 19%	251,850	+ 17%
Notched-Unnotched Ratio	1.16	0.92	-21%	1.10	+ 20%
Fracture Stress	Not Recorded	Not Recorded		296,900	
Elongation in 1/2" (4 Diameters), %	26	36	+ 38%	34	-5.6% NIL
Reduction of Area, %	36	43	+ 19%	23	-47%

TABLE 20 SUMMARY OF TEST RESULTS, A-286 STEEL (AMS 5737)

PROPERTY	TEST CONDITIONS				
	ROOM TEMP, UNIRRADIATED	30° R, UNIRRADIATED		30° R, IRRADIATED TO 1×10^{17} nvt FAST NEUTRONS	
	Test Data, Average of 5 Tests	Test Data, Average of 5 Tests	Net Change Due to Temperature	Test Data, Average of 3 Tests	Net Change Due to Irradiation
Ultimate Tensile Strength, F_{tu}, psi	171,300	246,700	+ 44%	232,000	-6%
Tensile Yield Strength, F_{ty}, psi	133,600	161,700	+ 21%	162,900	+ 0.7% NIL
F_{ty}/F_{tu} Ratio	0.78	0.66	-15%	0.70	+ 6%
Tensile Notch Strength ($K_t = 6$) psi	187,100	254,400	+ 36%	232,800	-8.5%
Notched-Unnotched Ratio	1.09	1.03	-5.5% NIL	1.00	-3% NIL
Fracture Stress	270,100	437,500	+ 62%	354,500	-19%
Elongation in 1/2" (4 Diameters), %	23	32	+ 39%	29	-9.4% NIL
Reduction of Area, %	52	45	-13%	42	-7% NIL

TABLE 21 SUMMARY OF TEST RESULTS, AM 350 STEEL, SCT

PROPERTY	TEST CONDITIONS				
	ROOM TEMP, UNIRRADIATED	30° R, UNIRRADIATED		30° R, IRRADIATED TO 1×10^{17} nvt FAST NEUTRONS	
	Test Data, Average of 5 Tests	Test Data, Average of 5 Tests	Net Change Due to Temperature	Test Data, Average of 3 Tests	Net Change Due to Irradiation
Ultimate Tensile Strength, F_{tu}, psi	199,300*	315,100	+ 58%	313,400	-0.5% NIL
Tensile Yield Strength, F_{ty}, psi	186,100*	314,200	+ 69%	309,800	-1.4% NIL
F_{ty}/F_{tu} Ratio	0.934	0.997	+ 6.7% NIL	0.989	-0.8% NIL
Tensile Notch Strength (K_t = 6) psi	256,700*	267,800	+ 4.3%	148,500	-44.5%
Notched-Unnotched Ratio	1.29*	0.85	-34%	0.47	-44.7%
Fracture Stress	314,400	490,400	+ 56%	377,700	-23%
Elongation in 1/2" (4 Diameters), %	16*	9.96	-38%	6.9	-31%
Reduction of Area, %	52*	36	-31%	20.6	-42.8%

*Value differs from previously reported figure (Quarterly Progress Report No. 13, ER 6929) due to additional testing to complete five-specimen sample lot.

TABLE 22 SUMMARY OF TEST RESULTS, 440C STAINLESS STEEL

PROPERTY	TEST CONDITIONS: ROOM TEMP, UNIRRADIATED	30° R, UNIRRADIATED		30° R, IRRADIATED TO 1×10^{17} nvt FAST NEUTRONS	
	Test Data, Average of 5 Tests	Test Data, Average of 5 Tests	Net Change Due to Temperature	Test Data, Average of 3 Tests	Net Change Due to Irradiation
Ultimate Tensile Strength, F_{tu}, psi	317,000	259,700	-18%	218,800	-15.7%
Tensile Yield Strength, F_{ty}, psi	272,800	259,700*	-4.8%	218,800*	-15.7%
F_{ty}/F_{tu} Ratio	0.86	1.00	+ 16.3%	1.00	NIL
Tensile Notch Strength (K_t = 6) psi	188,100**	103,000**	-45%	106,900	+ 3.8% NIL
Notched-Unnotched Ratio	0.59	0.40	-32%	0.49	+ 23%
Fracture Stress	343,200	259,700*	-24%	218,600*	-15.8%
Elongation in 1/2" (4 Diameters), %	5.2	NIL	-100%	NIL	NIL
Reduction of Area, %	8.6	NIL	-100%	NIL	NIL

*Sample failure without measurable plastic deformation or stress reduction equates these values with the ultimate tensile strength.

**Test data from reduced sample size lot. See Tables 17 and 18.

possible to show differences in micro-structure which arise from the combination of irradiation and cold work, even though differences due to radiation alone would not be observable. Also, qualitative differences in degree of deformation in one phase or another would be observable when phases are large and well delineated.

Sections from the fractured tensile specimens including necked and unnecked but elongated portions were mounted by standard techniques and surfaces were prepared in most cases by the techniques recommended by the vendor of the material. The irradiated samples were photographed in the hot-cell under somewhat different conditions from those for the unirradiated samples but these differences would not mask gross differences in surface appearances.

The metallographs of the fractured irradiated specimens were observed for effects due to irradiation and deformation at low temperature by comparing with metallographs of specimens fractured at room temperature and 30°R without irradiation. They were also compared with metallographs of the undeformed material supplied by the vendors and appearing in the Pedigree Report.[15]

In general it appears that the tensile measurements are much more sensitive than the metallographic observations to effects of radiation since rather large changes in reduction in area, fracture stress or elongation appear necessary before a corresponding qualitative difference is seen in the metallographs.

Precipitate (age hardenable) alloys which are of fairly simple structure and which have easily observable characteristics, dependent on the number and size of precipitates, are of particular importance in the general study of radiation effects in metals and alloys. The clusters of precipitated atoms in most of these alloys are the same order of magnitude as the supposed region affected by a neutron displaced atom and also of a size to affect the tensile properties.

Effects of neutron ~~radiations~~ at room temperature, even in pure metals, are actually like those of precipitation phenomenon in many respects and particularly with respect to mechanical properties.[16] In actual precipitation alloys, neutron irradiation at room temperature has been known to cause a re-solution of small precipitates at the same time causing

(15). LNP Pedigree Report, ER-5542 and Addendum.

(16). A. Seeger and J. Diehl, Properties of Reactor Materials and the Effects of Radiation Damage, pp 269, Butterworths, London 1962.

an acceleration of diffusion processes resulting in an opposing precipitation of solute atoms or growth of existing clusters.[17] At lower irradiation temperatures, the irradiation would be expected to cause a re-solution of small precipitates and the opposing diffusion process would not be as pronounced as at room temperature.[18]

It is difficult to predict how such processes should then effect the tensile properties without previously determining the properties of the materials as a function of measured precipitate size and distribution, but at least one can say that even at low temperatures the effect of irradiation might be similar to aging.

The results from Titanium 6% Al, 4% V, annealed and aged, which were previously published[19] indicated that the annealed (solution treated) material irradiated to 1×10^{17} nvt does have tensile characteristics approaching those of the unirradiated aged material fractured at the same temperature.

The most notable effect was a 53% decrease in the reduction in area in the annealed material due to previous irradiation which more than coincidentally corresponds to a 58% decrease in the reduction in area of the aged, but unirradiated, material.

Figure 1 shows the necked down region of Titanium 6% Al, 4% V annealed and unirradiated after fracture at 30°R. Figure 2 shows the necked down region of Titanium 6% Al, 4% V, annealed and irradiated, after fracture at 30°R.

A very pronounced difference in the types of fracture is observed in the two cases, which, reasonably, would, in some way be related to the large differences in reduction in area. The unirradiated material shows a jagged outline generally associated with intergranular fracture whereas the irradiated material shows a smooth outline generally associated with transgranular fracture.

(17). J. Denny, Bull. Am. Phys. Soc., Volume 29, pp 14-20 (1954).

A. Boltax, Radiation Effects on Materials - Vol. 1, ASTM Special Technical Publications No. 208, pp 183 (1956).

R. E. Jamison, Bull. Am. Phys. Soc., Ser. II, Vol. 2, pp 151 (1957) and Vol. 3, pp 118 (1958).

(18). M. S. Wechsler and R. H. Kernoham, Radiation Damage in Solids, Vol. II, pp 81, International Atomic Energy Agency, Vienna, 1962.

(19). Quarterly Progress Report No. 14, Pages 30 and 31.

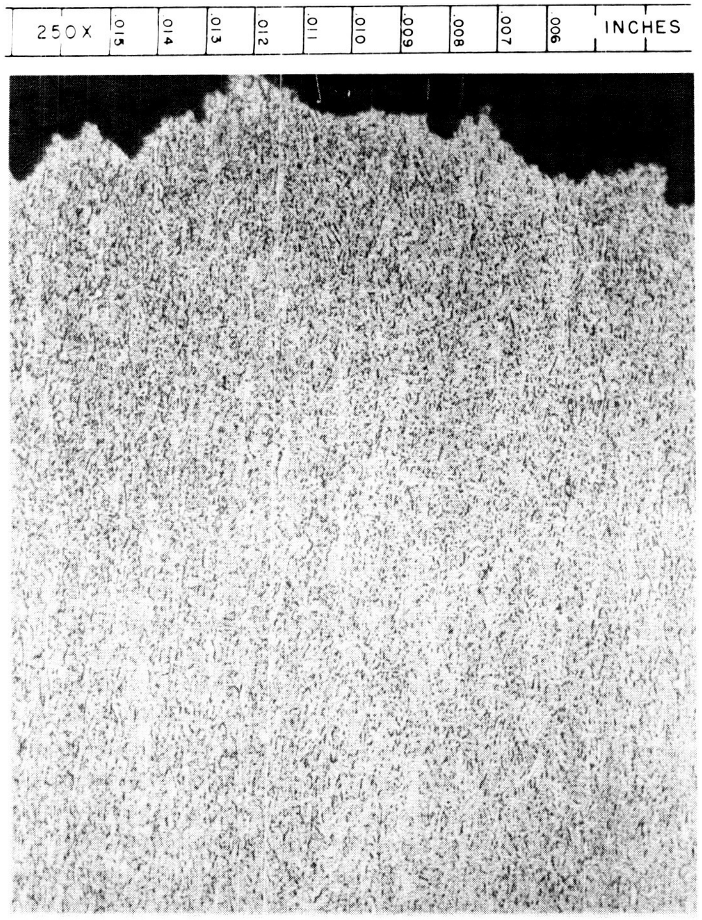

FIGURE 1 NECKED DOWN REGION OF TITANIUM 6% AL, 4% V, ANNEALED, UNIRRADIATED, AFTER FRACTURE AT 30° R, 250X

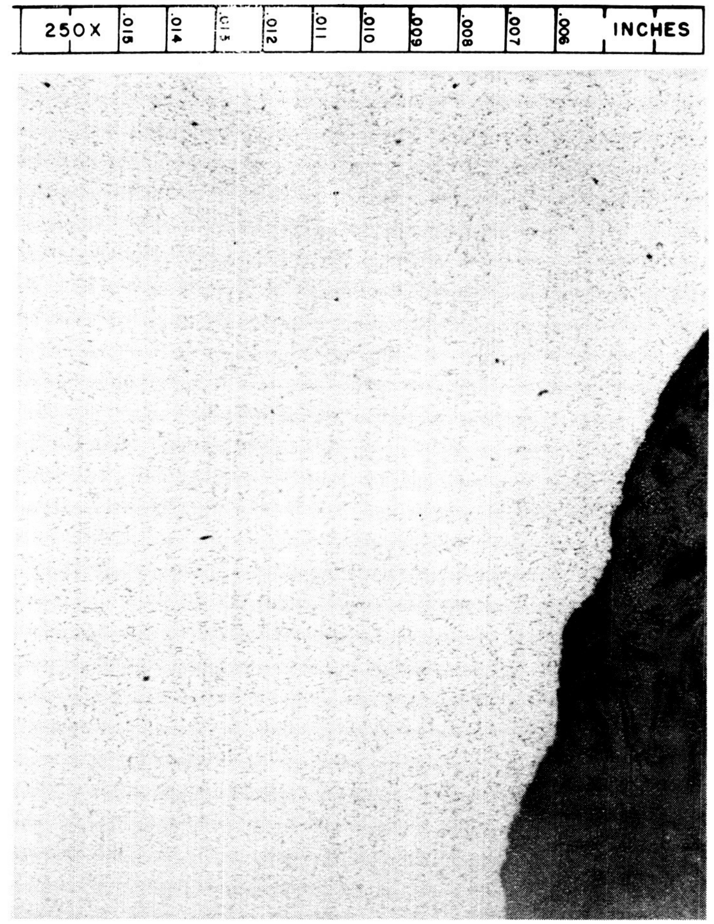

FIGURE 2 NECKED DOWN REGION OF TITANIUM 6% AL, 4% V, ANNEALED, AFTER IRRADIATION TO 1×10^{17} NVT AND FRACTURE AT 30° R, 250X

Titanium 6% Al, 4% V after aging and irradiation shows an increase in the ultimate strength due to radiation but no change in the reduction in area due to radiation or any other very large differences as in the case of the annealed (not aged) material. And, as might be expected, there are no obvious differences between the metallographs of the irradiated and unirradiated aged material.

The effect shown here might be explained in a number of ways but, without more knowledge of the starting grain size and precipitate size and distribution, conjecture should be limited. The metallographs can, in this case, be accepted merely as confirmation of the unusual tensile test results.

Electron micrograph studies would be particularly valuable for this material since they would show the actual precipitates and probably their re-distribution due to radiation. Nearly all structural materials and particularly steels strengthened with carbon can in a sense be described as involving precipitation related phenomena, and electron micrograph studies of any relatively simple precipitate alloy such as Titanium 6% Al, 4% V would contribute to the understanding of the radiation effects in the more complex structural materials.

Titanium 5% Al, 2.5% Sn with standard interstitial content also showed metallograph results corresponding with the tensile results. This material had a 34% decrease in the reduction in area along with a 20% decrease in fracture stress and a 12% decrease in elongation due to previous irradiation.

Figure 3 shows Titanium 5% Al, 2.5% Sn (standard interstitial), unirradiated, at the necked down region after fracture at 30^{o}R. Figure 4 shows Titanium 5% Al, 2.5% Sn (standard interstitial), irradiated, at the necked down region after fracture at 30^{o}R.

These photographs show that the break is more transgranular in the irradiated material, corresponding again to the large decrease in reduction in area. Close examination of the specimen shows there is also some apparent difference in the grain structure in the necked down region which does not appear in the photograph and it cannot be described as due to specific causes.

The same material, but with extra low interstitial content, showed some sizable differences in tensile properties due to irradiation with a larger effect on the elongation and less effect on the fracture stress than in the case of the standard interstitial content.

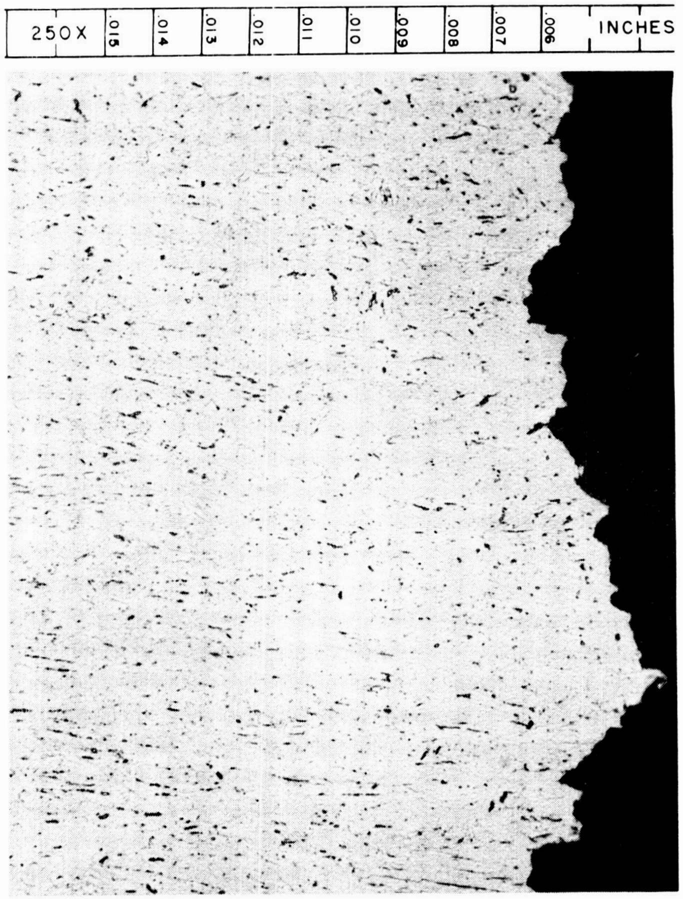

FIGURE 3 NECKED DOWN REGION OF TITANIUM 5% AL, 2.5% SN, STANDARD INTERSTITIAL CONTENT, UNIRRADIATED, AFTER FRACTURE AT 30° R, 250X

FIGURE 4 NECKED DOWN REGION OF TITANIUM 5% AL, 2.5% SN, STANDARD INTERSTITIAL CONTENT, AFTER IRRADIATION TO 1×10^{17} NVT AND FRACTURE AT 30° R, 250X

Figure 5 shows the necked down region of Titanium 5% Al, 2.5% Sn with extra low interstitials after irradiation and fracture at 30° R.

There is a large effect due to irradiation, appearing here that does not appear in the material with standard interstitial content. The necked down region shows large voids but less deformation of crystallites than in the standard interstitial material similarly irradiated and fractured. There is no apparent difference in the fracture. The voids seem to correspond to intergranular cracking brought on by unusual hardening of the crystallites.

Results of the metallographic studies are not entirely consistent, particularly in the case of 310 Stainless Steel, A-286 Alloy, and Titanium 8-1-1, and these inconsistencies point up the need for x-ray and electron micrograph studies to supplement the tensile test data.

FIGURE 5 NECKED DOWN REGION OF TITANIUM 5% AL, 2.5% SN, EXTRA LOW INTERSTITIAL CONTENT, AFTER IRRADIATION TO 1×10^{17} NVT AND FRACTURE AT 30° R, 250X

APPENDIX A

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